NON-PUBLIC?: N
ACCESSION #: 8905160294
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Quad Cities Unit One PAGE: 1 OF 5

DOCKET NUMBER: 05000254

TITLE: Manual Reactor Scram In Response to Erratic Main Turbine Control Valve and Bypass Valve Operation Due to Failure of Circuit Board Within the Electrohydraulic Control System
EVENT DATE: 04/12/89 LER #: 89-003-00 REPORT DATE: 05/05/89

OPERATING MODE: 4 POWER LEVEL: 064

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: TG COMPONENT: ECBD MANUFACTURER: G080
REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO EXPECTED SUBMISSION DATE:

ABSTRACT:

On April 12, 1989, Quad Cities Unit One was in the RUN mode at approximately 74 percent of rated core thermal power. At 1136 hours, a manual reactor scram was initiated due to main turbine bypass valves opening. One bypass valve had oscillated open during the night before, but at 1126 hours, all nine bypass valves had opened in sequence. NRC notification was completed at 1210 hours to comply with 10 CFR 50.72(b)(2)(ii).

An investigation revealed that the cause for this event was component failure. A circuit board within the combined maximum flow limit circuit had a decreasing output. The board limits the opening of control valves, and as a result of the decreasing output, caused the control valves to close. The bypass valves were opening as designed to control reactor pressure. The circuit board was replaced. This report is provided to satisfy 10 CFR 50.73(a)(2)(iv).

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END OF ABSTRACT

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION: Manual Reactor Scram In Response to Erratic Main Turbine Control Valve and Bypass Valve Operation
Due to Failure of Circuit Board Within the
Electrohydraulic Control System

A. CONDITIONS PRIOR TO EVENT:

Unit: One Event Date: April 12, 1989 Event Time: 1133
Reactor Mode: 4 Mode Name: RUN Power Level: 64%

This report was initiated by Deviation Report D-4-1-89-029.

RUN Mode (4) - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RBM interlocks in service (excluding the 15% high flux scram).

B. DESCRIPTION OF EVENT:

At 2321 hours, on April 11, 1989, Unit One was in the RUN mode under Economic Generation Control (EGC) at approximately 90 percent of rated core thermal power. At this time, the main turbine #1 bypass valve (BV) VI opened. The Nuclear Station Operator (NSO), as instructed by the Shift Control Room Engineer (SCRE), tripped the unit out of EGC and placed the reactor recirculation pumps AD!P! in manual control. This allowed the BV to close. At 2333 hours, the #1 BV came full open and the #2 BV oscillated up to 25 percent open. After recirculation pump speed was reduced to decrease reactor thermal power 2.4 percent, both BVs closed.

The on-call Instrument Maintenance Department (IMD) personnel and a Qualified Nuclear Engineer (QNE) were called in to the station. During the course of the next three and one-half hours, six load drops of approximately 2.4 percent of rated thermal power were used to close the

#1 BV each time it cycled open. A QNE verified the reactor was not the cause for the BV problem after evaluating computer trends and chart recorders for reactor pressure, turbine throttle pressure, Average Power Range Monitors (APRM) and reactor water level. Also, the QNE evaluated the Core Monitoring Code (CMC) edits before and after the event. At 0255 hours, on April 12, 1989, the IMD found an apparent problem with the bypass valve bias potentiometer. The BV potentiometer applies a bias to the bypass valve control circuitry to keep the valves closed during normal system pressure fluctuations. The potentiometer was intermittently giving zero voltage signals as opposed to a normal signal of negative 0.125 volts. As a result, no close signal was being sent to the BV. The IMD deduced that the potentiometer contacts were dirty at the normal setpoint. The IMD adjusted the potentiometer to increase the bias signal to the BVs. At 0305 hours, a load increase was started per the Load Dispatcher.

At 0310 hours, as reactor power was increased, the #1 BV opened. Another adjustment was made to the potentiometer and the BV closed. At 0312 hours, the #1 BV opened fully and the #2 BV opened 50 percent. Adjustments to the potentiometer were ineffective; therefore, reactor power was dropped to 68 percent of rated core thermal power, which allowed the #2 BV to go closed and the #1 BV to close halfway. At 0510 hours, the IMD made another adjustment to the potentiometer which caused the #1 BV to close. Additionally, the IMD set up strip charts and other monitoring equipment on the EHC system. At 0740 hours, load was increased per the Load Dispatcher.

At 1042 hours, with reactor power at 98 percent of rated core thermal power, the #1 BV cycled open. Reactor power was reduced and the BV closed. Again, the #1 BV cycled open and reactor power was reduced to allow the valve to close. Control rods were inserted in sequence to reduce reactor power. The BVs slowly began opening in sequence, and at 1112 hours, Unit One was at 74 percent of rated core thermal power with six BVs open. The IMD was still monitoring the signals in the EHC panel, but was unable to determine the cause for the BVs to be opening. At 1126 hours, all nine BVs were open. It was noted that the control valves (CV) were continuing to close and reactor pressure was increasing. Station personnel, with corporate concurrence, decided to manually scram the reactor at 1136 hours. When the turbine tripped, the bypass valves closed as expected and operated as designed to control reactor pressure. The expected water level transient due to the collapse of the voids following the scram caused reactor vessel level to drop below the +8 inches which caused Group II and III Primary Containment Isolations (PCI) JC!, Reactor Building Ventilation VA! and Control Room Ventilation VI! Isolations, and Standby Gas Treatment BH! initiation. Reactor water level was restored automatically by the Feedwater System JB! and a normal scram recovery proceeded. The Group II and III isolations were promptly reset. NRC notification using the Emergency Notification System (ENS) was completed at 1210 hours to comply with the requirements of 10 CFR 50.72(b)(2)(ii).

After the scram, the IMD began troubleshooting the EHC System under Work Request Q74995. A General Electric (GE) representative was brought in to assist in troubleshooting and concurred with the tests the station was utilizing.

Continuing the troubleshooting, the Pressure Amplifier, Load Control Unit, Control Valve Position Loops, and First Hit Box components were tested. Additionally, the EHC Lineup Functional Test was performed in

which several signals are provided as inputs to the EHC System to test the response. During the test, the CV position indicators on the 901-7 panel in the Control Room were oscillating, although the input to the system was a constant load signal. This result directed the IMD to the maximum combined flow limit circuit. Upon measuring the voltages at the different test points on the circuit boards within the maximum combined flow limit circuit, the IMD found that the A64 circuit board had a decreasing output. The A64 circuit board was replaced.

After replacing the circuit board, the maximum combined flow limit circuit was tested and responded properly.

At 0640 hours, on April 15, 1989, the Unit One NSO commenced reactor startup, and at 1051 hours, the reactor was critical.

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C. APPARENT CAUSE OF EVENT:

This event is being reported according to 10 CFR 50.73(a)(2)(iv), which requires the reporting of any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

The cause of this event was component failure. The circuit board is an original board and can be considered a random failure. The circuit board

(A64) within the maximum combined flow limit circuit was giving a decreasing output. The amplifier limits the opening of the CVs, and as a result of the decreasing output, was causing the CVs to drift closed. As the CVs drifted closed, the BVs opened as designed to control reactor pressure.

D. SAFETY ANALYSIS OF EVENT:

The safety significance for this event is minimal. All ESF actuations occurred as expected to bring the reactor to a safe shutdown condition.

At the time the reactor was scrammed, the reactor scrams that anticipate turbine trips were automatically bypassed because turbine first-stage pressure was less than 45 percent. Reactor power was approximately 74 percent of rated core thermal power. The function of these scrams is to anticipate the pressure, neutron flux and heat flux transient that would result from a rapid closure of the turbine stop valves or control valves. These scrams prevent the Minimum Critical Power Ratio (MCPR) safety limit from being exceeded assuming the turbine stop valves or control valves go closed with the BVs failing to open. A review of this condition was made at the time of the event, and it was determined that Technical Specifications were not being violated. GE was requested to analyze this situation. GE performed an analysis of a turbine trip without a stop valve closure scram from 90 percent thermal power and 40 percent bypass flow. GE concluded the expected change in Critical Power Ratio (CPR) from this event to be no greater than 0.15. This is significantly less than the calculated value of 0.21 for turbine trips without bypass and load rejection without bypass for Quad Cities Unit One Cycle 10 (GE-DRF No. J11-01007). Therefore, the operating conditions at the time of the scram were bounded by the results of previous analyses.

E. CORRECTIVE ACTIONS:

The immediate corrective action included replacement of the A64 circuit board.

The preventive maintenance program currently being developed for the EHC System will be examined for applicability to electronic components (NTS 2652008901903).

Additionally, the bypass valve opening bias potentiometer will be replaced during the next refuel outage (NTS 2542008902901).

This report will be covered under the lessons learned portion of license requalification training (NTS 2542008902902).

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F. PREVIOUS EVENTS:

There have been three previous events of a reactor scram being caused by a failure with the EHC System since 1980:

DVR 4-2-82-52 Reactor Scram From High Neutron Flux; found broken wire in power - load unbalance circuit causing
pressure spike.

DVR 4-2-87-65 Unit Two Reactor Scram Due to Failure of Turbine Master Trip Solenoid Valve.

DVR 4-2-89-019 Turbine Trip - Reactor Scram While Testing Turbine Master Trip Solenoid Valve.

G. COMPONENT FAILURE DATA:

The circuit board that failed was manufactured by General Electric, Model No. 942D3140. The Electro Hydraulic Control System (EHC) is not reportable in the Nuclear Plant Reliability Data System (NPRDS); therefore, the circuit board is not reportable.

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RLB-89-093

May 5, 1989

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Quad Cities Nuclear Power Station
Docket Number 50-254, DPR-29, Unit One

Enclosed is Licensee Event Report (LER) 89-003, Revision 00, for Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(iv): the licensee shall report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

R. L. Bax
Station Manager

RLB/AAF/ad

Enclosure

cc: R. Stols
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1938H

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